NON-PUBLIC?: N
ACCESSION #: 9211030022
LICENSEE EVENT REPORT (LER)

FACILITY NAME: PLANT HATCH, UNIT 1 PAGE: 1 OF 6

DOCKET NUMBER: 05000321

TITLE: MANUAL SCRAM DUE TO HIGH MAIN TURBINE VIBRATION CAUSED BY
COMPONENT FAILURE
EVENT DATE: 09/30/92 LER #: 92-024-00 REPORT DATE: 10/29/92

OTHER FACILITIES INVOLVED: PLANT HATCH UNIT 2 DOCKET NO: 05000366

OPERATING MODE: 1 POWER LEVEL: 086

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR SECTION:
50.73(a)(2)(i) and 50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:
NAME: STEVEN B. TIPPS, MANAGER NUCLEAR TELEPHONE: (912) 367-7851
SAFETY AND COMPLIANCE, HATCH

COMPONENT FAILURE DESCRIPTION:
CAUSE: X SYSTEM: SB COMPONENT: PDS MANUFACTURER: M235
REPORTABLE NPRDS: Y

SUPPLEMENTAL REPORT EXPECTED: NO

ABSTRACT:

On 9/30/92 at 0733 CDT, Unit 1 was in the Run mode at a power level of 2100 CMWT (86% rated thermal power). At that time, the unit was manually scrammed due to high turbine vibration. Reactor water level decreased as expected due to void collapse from the rapid decrease in power. This caused isolation of Groups 2 and 5 Primary Containment isolation valves, isolation of Unit 1 Secondary Containment, initiation of Units 1 and 2 Standby Gas Treatment systems, trip of both Reactor Recirculation pumps, and initiation of the High Pressure Coolant Injection and Reactor Core Isolation Cooling systems. All systems functioned per design except that four Unit 2 Secondary Containment isolation dampers did not close. Water level reached a minimum of 38 inches below instrument zero (120 inches above the top of the active fuel) before being recovered. Reactor pressure was controlled by the Turbine Bypass Valves at its pre-scram

value.

The cause of the high vibration was component failure. The bellows of pressure switch 1N38-N776 developed a leak which caused water to accumulate in its housing. The water shorted the switch terminal points causing the Reheat Steam Source Valves to the Moisture Separator Reheaters to close on a false low pressure signal. This caused a drop in the low pressure turbine steam inlet temperature and an increase in turbine vibration as different parts of the turbine cooled at different rates. The cause of the dampers not closing was a less than adequate design. The automatic isolation function of the dampers was defeated when their control switches were placed in the "open" position.

Corrective actions include replacing the pressure switch, tagging the damper control switches in the "auto" position, and implementing a design change.

END OF ABSTRACT

PLANT AND SYSTEM IDENTIFICATION

General Electric - Boiling Water Reactor
Energy Industry Identification System codes are identified in the text as (EIIS Code XX).

DESCRIPTION OF EVENT

On 9/30/92 at 0707 CDT, Unit 1 was in the Run mode at a power level of 2436 CMWT (100% rated thermal power). At that time, several feedwater heater (EIIS Code SJ) and Moisture Separator Reheater (MSR, EIIS Code SB) annunciators alarmed in the Main Control Room (EIIS Code NA). These annunciators included seventh stage feedwater heaters A and B low water level, MSR A and B drain tank low water level, MSR C and D drain tank low water level, and first stage MSR drain tank low water level. While investigating the cause for these alarms, Operations personnel noticed that the two Reheat Steam Source Valves (RSSVs, EIIS Code SB), 1N38-F101A and B, to the second stage of the MSRs were closed. Above 20% rated thermal power, the RSSVs should be open; therefore, Operations personnel attempted to open them with their control switch. They would not open.

Because of the reduction in reheat steam flow to the MSRs, feedwater temperature and feedwater heater pressures began to decrease slowly. Therefore, Operations personnel entered abnormal operating procedure 34AB-N21-001-1S, "Loss of Feedwater Heating," and began reducing power by

20% as required by the procedure. During the power reduction, Main Turbine (EIIS Code TA) vibration began to increase. At 1715 CDT, power had been reduced to 86% rated thermal power. At that time, power reduction was stopped as Operations personnel determined that a loss of feedwater heating had not occurred and that the alarms were the result of a loss of second stage MSR reheat steam supply. Main Turbine vibration levels continued to increase.

At 0730 CDT, vibration on Main Turbine bearing #3 reached 12 mils (its normal value is less than three mils). At 0733 CDT, with Unit 1 in the Run mode at 86% rated thermal power and vibration on bearing #3 still at 12 mils, the unit was manually scrammed in preparation for tripping the Main Turbine. This was done to reduce the reactor vessel pressure transient expected from a turbine trip.

Following the scram, reactor water level decreased as expected due to void collapse from the rapid decrease in reactor power. As water level decreased from its normal level of 37 inches above instrument zero (195 inches above the top of the active fuel) to 12.5 inches above instrument zero, another scram signal and a Group 2 Primary Containment Isolation System (PCIS, EIIS Code JM) isolation signal were received. The Group 2 PCIS valves closed on low (Level 3) reactor water level per design. This occurred approximately six seconds after the manual scram.

Water level decreased to its minimum value of 38 inches below instrument zero (120 inches above the top of the active fuel) about 13 seconds after the scram. This resulted in an isolation signal to the Group 5 PCIS, isolation of Unit 1 Secondary Contai
ment, which includes the Unit 2
Refueling Floor Secondary Containment isolation dampers, initiation of Units 1 and 2 Standby Gas Treatment (SBGT, EIIS Code BH) systems, a trip of both Reactor Recirculation pumps (EIIS Code AD), and initiation of the High Pressure Coolant Injection (HPCI, EIIS Code BJ) and Reactor Core Isolation Cooling (RCIC, EIIS Code BN) systems on low low (Level 2) reactor water level per design. All systems functioned as designed except that the four Unit 2 Refueling Floor Secondary Containment isolation dampers, 2T41-F003A and B and 2T41-F023A and B, failed to close. All other isolation valves and dampers closed within their allowable times. This was confirmed using the Safety Parameter Display System (EIIS Code IQ) computer tape of the event. Investigation revealed the four Secondary Containment isolation dampers did not close because their control switches had been placed in the "open" position, effectively defeating their automatic isolation logic, approximately seven hours prior to the manual scram. This rendered the dampers

inoperable in violation of the requirements of Unit 2 Technical Specifications section 3.9.5.2.

Water level was recovered using the "A" and "B" Reactor Feedwater Pumps (RFPs, EIIS Code SJ) and RCIC. HPCI started and ran on minimum flow, but did not inject to the vessel because water level was increased above the low low water level setpoint before its injection valve, 1E41-F006, received all its permissive signals to open. Water level increased to a maximum of approximately 58 inches above instrument zero. Operations personnel manually tripped the "B" RFP as water level increased. The "A" RFP, HPCI, and RCIC tripped on high reactor water level (Level 8) per design. The RFPs were restarted and used as necessary to maintain water at or above its normal level.

Reactor pressure was controlled by the Main Turbine Bypass Valves (EIIS Code SO) at its pre-scram pressure of 980 psig. No Safety Relief Valves lifted nor were any required to lift to control pressure.

CAUSE OF EVENT

The cause of the high Main Turbine vibration was component failure. The internal bellows of second stage MSR reheat steam supply pressure switch 1N38-N776 developed a small leak. The leak caused water to accumulate in the pressure switch's sealed housing over a period of time. The water shorted the switch terminal points causing the RSSVs, 1N38-F101A and B, to close on a false low reheat steam supply pressure signal. Closure of the RSSVs caused a rapid drop in the temperature of the low pressure turbine inlet steam. This resulted in movement of the low pressure turbine rotor in relation to the turbine shell as the two bodies of metal cooled at different rates. This caused rubbing in the shaft seal area and an increase in Main Turbine vibration.

The cause of the Unit 2 Refueling Floor Secondary Containment isolation dampers not closing was a less than adequate design. The automatic isolation function of the dampers could be defeated by placing their control switches to the "open" position. No other Secondary Containment isolation dampers have switches which function in this manner.

Approximately seven hours prior to the scram, following clearance restoration activities, a licensed Operator moved the control switches for dampers 2T41-F003A and B and 2T41-F023A and B from "auto" to "open." The Operator used procedure 34SO-T41-006-2S, "Refueling Floor Ventilation System," to align the Unit 2 Refueling Floor Ventilation system (EIIS Code VG) to its normal configuration following restoration of Clearance

2-92-1165. The procedure required that isolation dampers 2T41-F003A and B and 2T41-F023A and B be confirmed to be in the open position. Because the term "open" was capitalized in the procedure, the Operator thought the procedure required him to place the control switches for these dampers to the switch position labeled "open." He did this and the dampers opened as expected (open is the dampers' normal position). Due to an inadequate design, the automatic Secondary Containment isolation function of these dampers is overridden with the control switches in the "open" position. Hence, they did not close upon receipt of a Secondary Containment isolation signal during this event.

REPORTABILITY ANALYSIS AND SAFETY ASSESSMENT

This report is required by 10 CFR 50.73(a)(2)(iv) because an unplanned manual actuation of the Reactor Protection System (EIIS Code JC) occurred. Specifically, the reactor was manually scrammed in response to high vibration of the Main Turbine. Subsequent to the manual scram, several Engineered Safety Feature systems automatically actuated per design because of a low reactor water level condition. Four Unit 2 Secondary Containment isolation dampers, however, did not close as required because their control switches had been mispositioned several hours prior to the scram. This rendered the dampers inoperable in violation of the requirements of Unit 2 Technical Specifications section 3.9.5.2. Consequently, this report is also required by 10 CFR 50.73(a)(2)(i) because a condition prohibited by the Unit 2 Technical Specifications existed.

High turbine vibration indicates a problem with the Main Turbine. If the high vibration is allowed to continue, severe damage to the turbine can result; therefore, it is important to reduce vibration quickly (e.g., by tripping the turbine). In this event, a manual scram was inserted prior to tripping the Main Turbine as a matter of good operating practice. Insertion of the manual scram precluded an automatic scram, which would have resulted from tripping the Main Turbine and resultant closure of the Main Turbine Stop Valves at greater than 30% rated thermal power.

As expected, reactor vessel water level decreased due to void collapse following the manual scram. This resulted in several Engineered Safety Feature systems automatically actuating because of a low reactor water level condition. Except as explained previously, all systems functioned as designed. Water level was never lower than 120 inches above the top of the active fuel. The Main Turbine Bypass Valves operated per design to maintain reactor pressure at its pre-scram value of 980 psig. No Safety Relief Valves actuated nor were any required to actuate.

Secondary Containment is designed to minimize any ground level release of radioactive material which may result from an accident. Upon receipt of various signals, e.g., low low reactor water level, which may indicate an accident has occurred, the Secondary Containment isolation dampers will automatically close and the SBGT systems will automatically start. These actions result in a negative pressure being drawn on the Secondary Containment by the SBGT system fans to minimize any untreated leakage of radioactive material from the Secondary Containment and to ensure the Secondary Containment atmosphere is filtered through the SBGT filters prior to its elevated release from the Main Stack (EIIS Code VL).

In this event, four Unit 2 Refueling Floor Secondary Containment isolation dampers failed to close on a Unit 1 Secondary Containment isolation signal as required. The dampers failed to close because their control switches had been mispositioned, effectively defeating their isolation logic. However, a review of the chart from the Secondary Containment differential pressure recorder indicated that the SBGT system fans were able to maintain the Secondary Containment under a negative pressure during this event even with the dampers open. Had an actual leakage of radioactive material occurred into the Secondary Containment, the SBGT system would have created adequate negative pressure in the Secondary Containment to minimize any untreated, ground level release of radioactive material per design.

Based on the above discussion, it is concluded that this event had no adverse impact on nuclear safety. This analysis is applicable to all power levels.

CORRECTIVE ACTIONS

Pressure switch 1N38-N776 was replaced with a new pressure switch on 9/30/92 per Maintenance Work Order 1-92-4563.

A clearance was placed on the control switches for dampers 2T41-F003A and B and 2T41-F023A and B to administratively maintain the switches in the "auto" position. This was done on 10/1/92. It was determined that a similar problem did not exist for the Unit 2 Reactor Building Secondary Containment and the Unit 1 Refueling Floor and Reactor Building Secondary Containment isolation dampers because these dampers do not have control switches.

A design change will be implemented to remove the ability to override the automatic isolation function for dampers 2T41-F003A and B and 2T41-F023A

and B when they are in the open position. Until the design change can be implemented, appropriate administrative controls will remain in place to ensure the dampers remain operable.

ADDITIONAL INFORMATION

No system other than those mentioned in this report were affected by this event.

Component Failure Information:

Master Parts List Number: 1N38-N776
Manufacturer: Mercoid Corporation
Model Number: DPSW-7233-153-64
Type: Differential Pressure Switch
Manufacturer Code: M235
EIIS System Code: SB
Reportable to NPRDS: Yes
Root Cause Code: X
EIIS Component Code: PDS

There have been two previous similar events reported in the last two years in which the RSSVs either closed prematurely or failed to close completely resulting in high Main Turbine vibration and a reactor scram. These events were reported in Licensee Event Reports 50-321/1990-020, dated 10/26/90, and 50-321/1990-021, dated 11/8/90. In the first event, one of the RSSVs did not close completely causing unequal heating of the low pressure turbines and high turbine vibration. The torque switch for this valve was not adjusted to allow it to close completely. In the second event, the RSSVs closed with the unit operating at 100% power. The most likely cause of that event was the mispositioning of the switch used for the steam blanketing mode for the MSRs. Corrective actions for these events included adjusting an RSSV torque switch and removing the steam blanketing mode switch. These corrective actions would not have prevented this event because they could not have prevented the failure of pressure switch 1N38-N776.

There have been no previous similar events reported in the last two years in which an inadequate design resulted in an Engineered Safety Feature system not actuating upon receipt of a trip signal.

ATTACHMENT 1 TO 9211030022 PAGE 1 OF 1

Georgia Power Company
40 inverness Center Parkway
Post Office Box 1295

Birmingham, Alabama 35201
Telephone 205 877-7279

J. T. Beckham, Jr. Georgia Power
Vice President - Nuclear Hatch Project the southern electric system

HL-2999
004208

October 29, 1992

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555

PLANT HATCH - UNITS 1, 2
NRC DOCKETS 50-321, 50-366
OPERATING LICENSES DPR-57, NPF-5
LICENSEE EVENT REPORT
MANUAL SCRAM DUE TO HIGH MAIN TURBINE VIBRATION
CAUSED BY COMPONENT FAILURE

Gentlemen:

In accordance with the requirements of 10 CFR 50.73(a)(2)(i) and 10 CFR 50.73(a)(2)(iv), Georgia Power Company is submitting the enclosed Licensee Event Report (LER) concerning a Unit 1 manual scram due to high main turbine vibration caused by component failure.

Sincerely,

J. T. Beckham, Jr.

MCM/cr

Enclosure: LER 50-321/1992-024

cc: Georgia Power Company
Mr. H. L. Sumner, General Manager - Nuclear Plant
NORMS

U.S. Nuclear Regulatory Commission, Washington, D.C.
Mr. K. Jabbour, Licensing Project Manager - Hatch

U.S. Nuclear Regulatory Commission, Region II
Mr. S. D. Ebneter, Regional Administrator

Mr. L. D. Wert, Senior Resident Inspector - Hatch

*** END OF DOCUMENT ***